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**RECORD OF ISSUE/REVISIONS**

<b>ISSUE AUTHORIZATION DATE</b>	<b>EFFECTIVE DATE</b>	<b>REV. NO.</b>	<b>DESCRIPTION</b>
Draft	07/14/2003	00-A	New document to establish the technical basis for the development of a radiation exposure matrix for Huntington Pilot Plant. Initiated by Jeri L. Anderson.
Draft	10/15/2003	00-B	Draft revision to incorporate NIOSH review comments. Initiated by Jeri L. Anderson.
Draft	10/22/2003	00-C	Draft revision to incorporate NIOSH review comments. Initiated by Jeri L. Anderson.
10/31/2003	10/31/2003	00	First approved issue. Initiated by Jeri L. Anderson. Restricted use – This TBD cannot be used to process cases that involve significant exposures to electrons (e.g., skin cancers).

**Basis for Development of an Exposure Matrix for  
HUNTINGTON PILOT PLANT**  
Huntington, West Virginia  
Period of Operation: 1951-1963

**1.0 Site Description, Operational History, and Process**

According to reports, the Huntington Pilot Plant (also known as the Reduction Pilot Plant) was built by the Atomic Energy Commission (AEC) in 1951 to supply nickel powder used to make gaseous diffusion barriers for the gaseous diffusion plants in Paducah, Kentucky, and Portsmouth, Ohio (Clark and Cottrell 1980, Berger 1981, US DOE 2002). A 1957 letter supplement also indicates that the AEC's operating contractor receiving the nickel powder was Union Carbide Nuclear Company (Humphries 1957). A facility data report (U.S. AEC 1955) and an International Nickel Company (INCO) memorandum (Keller 1957) both suggest that the powder was shipped to K-25 for barrier production.

An additional document (DOEIC 2000a) gave details of starting material and barrier material transactions of the Oak Ridge Gaseous Diffusion Plant (ORGD) for fiscal years 1959 through 1961. According to this document, during this time period ORGD shipped between 1,980,000 and 2,587,000 pounds of starting material to INCO per year. A narrative attached to the document indicates that these quantities were the required amounts that satisfied ORGD's portion of the total quantity of starting material that was to be shipped to INCO by the three gaseous diffusion plants. The amounts required by INCO were based on the assumption of receipts of powder from INCO at the rate of 390,000 pounds per month.

A memorandum survey report (White 1950) indicates that in January 1950, the AEC was considering awarding a contract to the International Nickel Company to melt K-25 scrap at their plant in Huntington, West Virginia. The work was to be performed in Furnace No. 5 in the Refinery Building. According to the survey, a railroad siding adjoined the Refinery Building, which would make it possible to unload and weigh the scrap at the siding, place in buckets while it was still in the sealed cartons, and charge the furnace with the sealed cartons. The memorandum also suggested that the intended schedule was to use the furnace for melting K-25 scrap for two weeks each month and that the melt period took eight hours. At the time of the memorandum survey report, it was estimated that about one million pounds of nickel scrap was waiting to be melted (White 1950).

In a letter supplement to Contract No. AT-(40-1)-1092, the AEC wrote to confirm that the INCO was willing to produce up to eight tons of metallic nickel powder with the required specifications (Humphries 1957). There were two sources of feed material for the nickel powder production. One source was uncontaminated nickel sinter. The other source was the above mentioned nickel scrap from the K-25 plant. Reports stated that this nickel scrap was contaminated with uranium with a maximum enrichment of 4% (by weight) (Clark and Cottrell 1980, Berger 1981). No information could be found regarding analyses performed to determine the extent of the uranium contamination. However, a recently declassified document entitled "Appendix 'C' – Specifications For Starting Material" (DOEIC 2000b) detailed the required minimum and maximum constituents for the nickel-containing barrier material. These specifications are shown in Table 1.

The Huntington Pilot Plant was shut down in 1963 and maintained in stand-by condition by Huntington Alloys, Inc., a subsidiary of INCO. The Department of Energy (DOE) performed a preliminary radiological survey of the site and recommended that the site be decontaminated. The decontamination of the facility took place between November 27, 1978 and May 18, 1979. According

Table 1. Specifications for nickel-containing barrier material for use in production of nickel powder by the carbonyl process.

Constituent	Maximum	Minimum
Nickel		98.0%
Water	0.50%	--
Iron	0.40%	--
Uranium-235	0.0875 grams per pound of starting material average in any one drum	--
Total uranium	500 ppm	--

to reports, the classified and contaminated scrap was transported to the Portsmouth Plant in Ohio in 59 truckloads and 4 railcar loads. There were 138 truckloads of clean scrap that were removed by the Cleveland Wrecking Company (Clark and Cottrell 1980). The only original building remaining on the site was the compressor building, which is a concrete block building consisting of large open storage area, a change room facility with an office upstairs, and a small electrical craft training room. The site was surveyed again in May 1979 and recommended for unrestricted release by the DOE.

A review of the May 1979 survey results by the Health and Safety Research Division of Oak Ridge National Laboratory (ORNL) revealed certain areas that required further evaluation. On August 13, 1980, ORNL performed a preliminary survey at the site to determine the need for additional surveys (Clark and Cottrell 1980). ORNL recommended a detailed formal survey that was performed by Oak Ridge Associated Universities' (ORAU) Radiological Site Assessment Program on November 17-19 and December 3, 1980 (Berger 1981). The results of the formal survey showed that the radiation levels at the site were within guidelines for unrestricted release of the property.

In the 1980 formal survey, radionuclide (U-238, U-235, and Ra-226) and nickel concentrations were measured in soil samples collected both onsite and offsite. The average concentrations of U-238 and U-235 in the onsite soil samples were 21 and 0.40 pCi/g of soil, respectively. The average concentrations of U-238 and U-235 in offsite soil samples were 4.5 and 0.06 pCi/g, respectively. Nickel concentrations were also measured in the soil samples. The average nickel concentration onsite was 8.2 mg/g of soil and the average concentration offsite was 3.5 mg/g of soil.

Results of the survey of the compressor building were compared with U.S. NRC release guidelines (US NRC 1976) and American National Standard Institute (ANSI) release standards (ANSI 1978) for uranium and its decay products. The comparison showed that measured levels were below the guidelines for both transferable and total alpha and beta-gamma contamination. External beta-gamma radiation dose rates at 1 cm from building surfaces were also below NRC guidelines. The NRC and ANSI documents provided no guidelines for gamma exposure levels. However, assuming a continuous (168 hours/week) exposure to the highest levels measured inside the compressor building resulted in an annual dose of less than 305 mrem. The same exposure time to outdoor gamma radiation levels would result in an annual dose of less than 200 mrem.

## 2.0 Estimation of Internal Exposure

No records were found of radioactivity monitoring during the period the K-25 scrap nickel was melted at the Huntington Plant. Thus, it was necessary to make a reasonable estimate of uranium air concentration and area contamination in order to estimate exposure to workers during the period that the plant operated under contract with the AEC.

International Nickel Company used the carbonyl process to produce high-purity nickel powders. The highly selective process involves exposing the melted nickel containing contaminants to carbon

monoxide gas. This results in the formation of nickel carbonyl gas, which is siphoned off while the contaminants remain behind in the residue. The nickel carbonyl gas is decomposed at moderate temperatures producing extremely pure nickel, and carbon monoxide that is recycled through the process. Figure 1 is a conceptual diagram of the nickel carbonyl process.

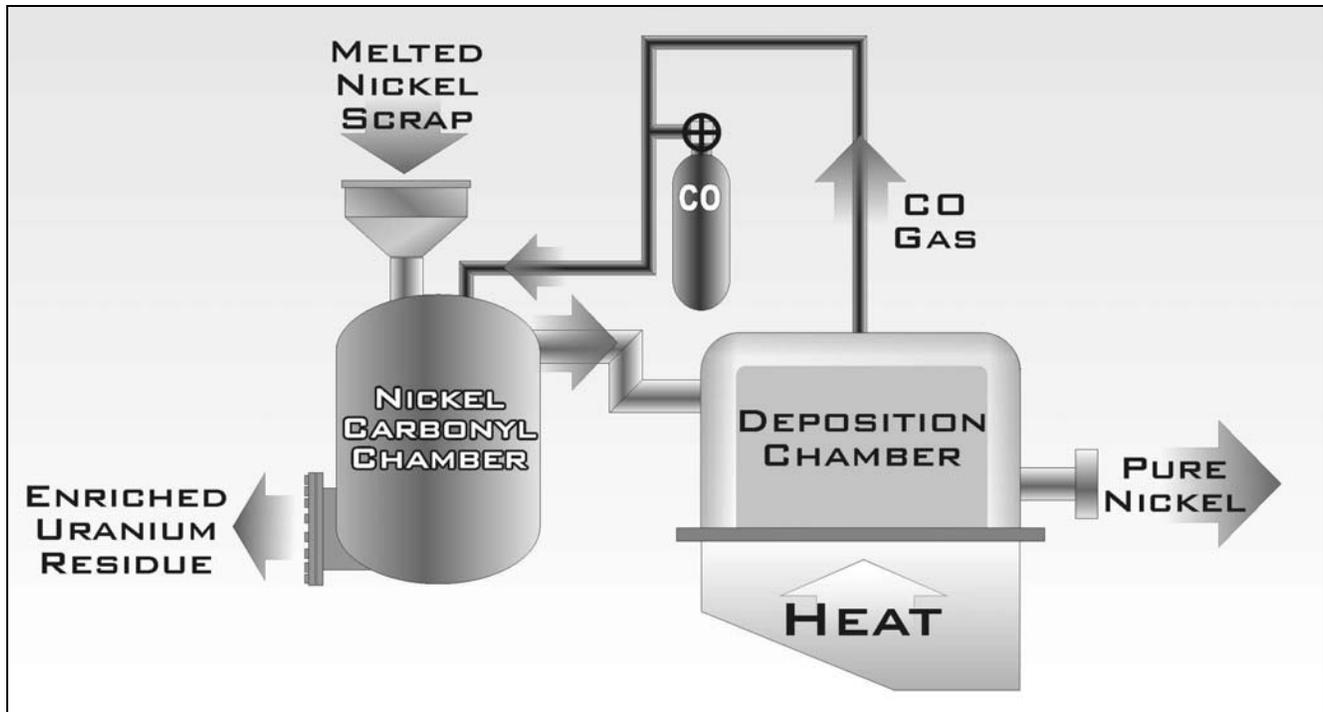


Figure 1. Conceptual diagram of the nickel carbonyl process. (Illustration courtesy of S. Jensen)

The scrap was supposedly shipped to INCO in cartons that remained sealed during weighing and charging of the furnace, the potential for exposure prior to the melting process was considered to be low. The potential for external exposure is highest in the area where the melted scrap is stored and where the residues from the carbonyl process are disposed of. Information regarding the disposition of the carbonyl process residues appears to remain classified.

According to the required specifications for the starting material, the maximum enrichment allowed was approximately 39%. However, most reports indicate that the maximum enrichment was ~4%. To be claimant favorable, it was assumed the uranium contaminating the nickel scrap was enriched to 39% by weight. Table 2 shows the composition of the uranium isotopes in uranium enriched to 39%.

Table 2. Composition of uranium isotopes in enriched uranium (maximum enrichment of 39%).

Isotope	Weight percent	Activity percent
U-234	0.3471	94.9106
U-235	39.0000	3.6914
U-236	0.1794	0.5102
U-238	60.4735	0.8877

Table 3 shows the calculated ratio of the uranium isotopes to nickel in the starting material based on the required specifications and assuming an enrichment of 39%.

Table 3. Ratios of uranium isotopes to nickel in starting material.

Uranium isotope	Ratio to Ni (pCi/mg Ni)
U-235	0.425
U-234	10.901
U-238	0.102

Documents giving details of starting material shipped from ORGDP to INCO in 1950-1961 indicated that annually between 1,980,000 and 2,587,000 pounds of contaminated nickel scrap could have been melted and used to produce nickel powder. Thus, assuming all starting material was contaminated with 39% enriched uranium, it is estimated that a residue containing approximately 0.37-0.49 Ci of U-235, 9.6-13 Ci of U-234, and 0.09-0.12 Ci of U-238 could have been produced per year. It is important to note that these quantities for nickel scrap are only the amounts reported by ORGDP. It is assumed that both the Paducah and Portsmouth gaseous diffusion plants contributed scrap material, although the quantities are not known at this time.

Workers in nickel smelting operations are exposed to mean nickel aerosol concentrations ranging from 0.01 to less than 3.0 mg Ni/m<sup>3</sup>. Mean nickel aerosol concentrations for refining operations range from 0.003 to about 1.5 mg Ni/m<sup>3</sup> (NiPERA 1997). The nickel concentration in air at the Huntington Plant was estimated from exposure information collected during a nickel worker mortality study. Table 4 shows estimates of nickel exposures at the plant.

These exposure estimates, made by International Nickel Company for each of the various departments, were intended to represent average airborne concentrations of nickel in all forms over an 8-hour period. Generally, modern exposure data were used to estimate historic exposures and, whenever possible, were adjusted on the basis of process changes and environmental controls that were implemented over the years. For some departments, historical sample data were available that were obtained by the midget impinger particle counting technique. These data were then converted to the modern gravimetric expression (Enterline and Marsh 1982).

According to Table 4, air concentrations of nickel at the Huntington plant ranged from 0.01 to 5.0 mg Ni/m<sup>3</sup> while the air concentration in the Reduction Pilot Plant was estimated at 0.02 mg Ni/m<sup>3</sup>. These measured values appear to be lognormally distributed with a median of 0.05 mg Ni/m<sup>3</sup> and a geometric standard deviation (GSD) of 4.3. To be claimant favorable, median (0.05 mg Ni/m<sup>3</sup>) of the values in Table 4 is used in combination with the ratios of U-235, U-234, and U-238 to nickel calculated from the required specifications (see Table 3) to create a lognormal distribution that can be used to estimate worker exposure to enriched uranium.

Table 5 shows the calculation of the annual intake of 39% enriched uranium from the estimated annual internal exposure to nickel dust contaminated with enriched uranium. To obtain these estimates, the ratio of enriched uranium to nickel is multiplied by the median nickel concentration in air to estimate a median enriched uranium air concentration. This air concentration is multiplied by the breathing rate and the number of hours worked to obtain a median annual intake. The breathing rate was calculated from the volume of air breathed for an adult light worker shown in ICRP Publication 66, Table 6 on pg. 23 (ICRP 1994). This category assumes an activity distribution of 1/3 sitting and 2/3 light exercise.

It was assumed that the worker was exposed to the contaminated nickel dust for a 2000-hour work year. Because the exact dates of operation are unknown, to be claimant favorable, it is assumed the the plant operated from January 1951 through December 1963. A survey report suggested that the

Table 4. Estimates of historical<sup>a</sup> nickel exposures at the International Nickel Company plant in Huntington, WV (reproduced from Enterline and Marsh 1982).

Department	Exposure (mg Ni/m <sup>3</sup> )
Acid reclaim	0.02
Blacksmith shop	0.02
Carpenter shop <sup>b</sup>	0.02
Chipping and hammer	0.75 <sup>c</sup>
Cold drawing (acid reclaim 1970 on)	0.10 <sup>c</sup>
Combustion <sup>b</sup>	0.05
Electrical <sup>b</sup>	0.10
Extrusion	0.20 <sup>c</sup>
General offices <sup>b</sup>	0.01
Heat treatment	0.02
Machine shop	0.03 <sup>c</sup>
Mechanical <sup>b</sup>	0.01
Melting and casting	0.25 <sup>c</sup>
Merchant mill	0.30 <sup>c</sup>
Night superintendent <sup>b</sup>	0.01
Pipe shop <sup>b</sup>	0.08
Polishing	0.10 <sup>c</sup>
Primary mill	0.06
Reduction pilot plant	0.02
Refinery	5.00 <sup>c</sup>
Research and development <sup>b</sup>	0.05
Roll grinding	0.03
Roll turning	0.03
Sheet mill (acid reclaim to 1970)	0.20 <sup>c</sup>
Shipping	0.01
Standards or industrial engineering <sup>b</sup>	0.01
Steel shop <sup>b</sup>	0.12
Stocks	0.01
Stores	0.01
Strip mill	0.20 <sup>c</sup>
Transportation <sup>b</sup>	0.01
Vacuum melting	0.15
Welding products	0.02
Yard <sup>b</sup>	0.10
Warehouse	0.01
Watchman <sup>b</sup>	0.01
Inspection (unassigned) <sup>b</sup>	0.05

- a. Based on current measurements except as noted.
- b. Plant-wide work assignment.
- c. Based on historic midget impinger counts.

melting of the contaminated nickel scrap was scheduled for two weeks per month (White 1950). Also, other sources of uncontaminated nickel were used for producing the metallic nickel powder (Clark and Cottrell 1980). Thus, this assumption is considered to be claimant favorable.

Table 5. Estimated annual internal exposure to nickel dust contaminated with 39%-enriched uranium.

	<b>Median</b>
Ni Air Concentration (mg/m <sup>3</sup> )	0.05
EU Air Concentration based on Ni Air Concentration (pCi/m <sup>3</sup> )	0.58
Breathing Rate (m <sup>3</sup> /h)	1.2
Hours Worked (h)	2,000
Annual Intake (pCi)	1,400

In order to make use of the information in Table 5 for estimation of organ dose from internal exposure, the annual doses for the organ of interest should be calculated in the IMBA NIOSH program for the median intake of U-234 assuming a chronic exposure. The calculated annual organ doses are then entered into the NIOSH IREP program as the annual dose due to a chronic exposure to alpha radiation using a lognormal distribution with a GSD of 4.3.

The chemical form of the uranium contaminating the nickel dust is unknown. Many details of gaseous diffusion technology remain classified such as the chemical effect of UF<sub>6</sub> on various components of gaseous diffusion plants (US DOE 1994). Prior to the enrichment by gaseous diffusion, the uranium compound is converted to UF<sub>6</sub>, which when exposed to moisture, breaks down to form HF and UO<sub>2</sub>F<sub>2</sub>. The HF most likely reacts with the nickel to form nickel fluorides. Other possible compounds are U, UF<sub>3</sub>, UF<sub>4</sub>, and UO<sub>2</sub> (Schneider 2003, Zhu 2003). UO<sub>2</sub>F<sub>2</sub> exhibits behavior consistent with Type F absorption (ICRP 1995, Schneider 2003). Human and animal studies suggest that UO<sub>2</sub> should be assigned to Absorption Type S. Uranium fluorides exhibit more complex behavior with some studies indicating assignment to Type F and some studies indicating assignment to Type M. The ICRP in Publication 71 recommends the use of Absorption Type M in the absence of specific information (ICRP 1995). The ICRP Publication 66 (ICRP 1994) default values should be used for the deposition parameters in the IMBA NIOSH program.

### **3.0 Estimation of External Exposure**

For estimating maximum external exposure due to submersion in air containing nickel dust contaminated with enriched uranium, the maximum air concentration values and work times per year from Table 5 were combined with dose coefficients for U-235, U-234, and U-238 and its daughter radionuclides Th-234 and Pa-234m from Federal Guidance Report No. 12 (US EPA 1993). The median annual organ dose estimates due to submersion in air for 1951 through 1963 were calculated. The cumulative maximum organ dose for 1951-1963 is less than 1 mrem and is therefore not included in the dose estimation.

To estimate external dose due to penetrating radiation from contaminated surfaces in the Reduction Pilot Plant, the estimated air concentration was multiplied by the indoor deposition velocity and the assumed exposure time. The indoor deposition velocity is dependent on the physical properties of the room (air viscosity and density, turbulence, thermal gradients, surface geometry) and the particles (diameter, shape, density). These characteristics are unknown, thus the terminal gravitational settling velocity of 0.002 m/s was used as an estimate of the velocity of deposition to surfaces in the building (CRC 1986). This settling velocity is for 5-µm particles with a density of 2.0 g/cm<sup>3</sup> in air at 25°C and 1 atm. This value falls in the range of deposition velocities (2.7E-06 to 2.7E-03 m/s)

measured in various studies (US NRC 2002) and is considered a conservative estimate. Also, room air exchange rates, ventilation, and plant housekeeping practices are unknown, so it is assumed that there is a steady state air concentration and surface contamination. The amount of surface contamination is assumed to be due to a 10-year buildup.

The estimated activity of the surface contamination is multiplied by the dose coefficients for contaminated ground surfaces for U-238, U-234, U-235, Th-234, and Pa-234m from Federal Guidance Report No. 12 (US EPA 1993). The annual organ dose estimated by this method is less than 1 mrem and is therefore not used for the photon dose estimation. Instead, survey data obtained from one of the remaining Huntington Pilot Plant structures (Berger 1981) is used to estimate external photon dose.

According to the 1981 survey report, only the compressor building remained at the site. Gamma radiation levels measured at 1 meter above the floor ranged from 9-12  $\mu\text{R/h}$  with a maximum reading of 35  $\mu\text{R/h}$ . To give claimant favorable estimate of external photon exposure during operations, it was assumed that the median exposure rate was 12  $\mu\text{R/h}$  and the upper 95<sup>th</sup> percentile was 35  $\mu\text{R/h}$ , giving a GSD of 1.9. Thus, to estimate potential external exposure to contaminated surfaces in the plant, the median exposure rate was multiplied by the Exposure (R) to Organ Dose (rem) photon dose conversion factors from Appendix B of the NIOSH External Dose Reconstruction Implementation Guideline (DHHS 2002). The exposure geometry was assumed to be isotropic and the exposure rate was divided evenly between the conversion factors for photons with energy between 30 and 250 keV and photons with energy greater than 250 keV. Table 6 shows the calculated annual organ doses from exposure to contaminated surfaces during plant operations.

Table 6. Annual organ doses due to exposure to surfaces contaminated with enriched uranium.

Organ	Annual organ dose (rem)		
	Photons E=30-250 keV	Photons E>250 keV	Total
Bladder	6.43E-03	7.76E-03	1.42E-02
Red bone marrow	6.68E-03	7.99E-03	1.47E-02
Bone surface	1.13E-02	8.17E-03	1.94E-02
Breast	8.50E-03	8.89E-03	1.74E-02
Colon	6.18E-03	7.61E-03	1.38E-02
Esophagus	6.00E-03	7.85E-03	1.38E-02
Eye	8.90E-03	9.11E-03	1.80E-02
Ovaries	5.93E-03	7.51E-03	1.34E-02
Testes	7.58E-03	8.32E-03	1.59E-02
Liver	6.82E-03	7.98E-03	1.48E-02
Lung	7.50E-03	8.47E-03	1.60E-02
Remainder organs	6.66E-03	7.96E-03	1.46E-02
Skin	8.77E-03	9.11E-03	1.79E-02
Stomach	6.79E-03	7.97E-03	1.48E-02
Thymus	7.37E-03	8.18E-03	1.56E-02
Thyroid	7.67E-03	8.57E-03	1.62E-02
Uterus	5.82E-03	7.25E-03	1.31E-02

It was also assumed that there was a potential to receive a shallow dose from electrons due to skin contamination. The amount of skin contamination was calculated by using a measured deposition velocity to skin of 0.012 m/s (Andersson et al. 2002, Fogh et al. 1999). For simplification, it was assumed that the material deposited on the skin during an 8-hour period was all deposited at the beginning of the shift, and the worker took a shower at the end of the shift. To determine the electron

shallow dose, the quantities of the minimum, mode, and maximum skin contamination was combined with electron dose-rate conversion factors for U-238, U-234, U-235, Th-234, and Pa-234m for skin in contact with radionuclides (Kocher and Eckerman 1987). This resulted in an annual skin dose of  $4.6E-06$  rem. This value is attributed to electrons ( $E > 15$  keV) in the NIOSH IREP program using a chronic exposure and a lognormal distribution with a GSD of 4.3.

Another potential for exposure existed in a worker's proximity to the carbonyl process residue, which contained an unknown quantity of enriched uranium. This process residue was most likely decanted into storage units known as "birdcages." Figure 2 below is an illustration of a birdcage showing the dimensions of the container as well as a 5-by-5 double-stacked array of birdcages. It was assumed that basic nuclear safety parameters were utilized at the plant to prevent the occurrence of a criticality accident. There are three general methods of control for preventing nuclear accidents: mass control, geometry control, and concentration control. According to an AEC report (US AEC 1961), the recommended maximum mass of U-235 in a container is 350 g (mass control) and the maximum diameter of infinite cylinder is 5.0 inches (geometry control). These parameters are based on moderation by water and were most likely used in the event of flooding of the storage area or fire. During the carbonyl process, it was possible to control the mass of U-235 by limiting the amount of contaminated scrap that was introduced into the system at one time.

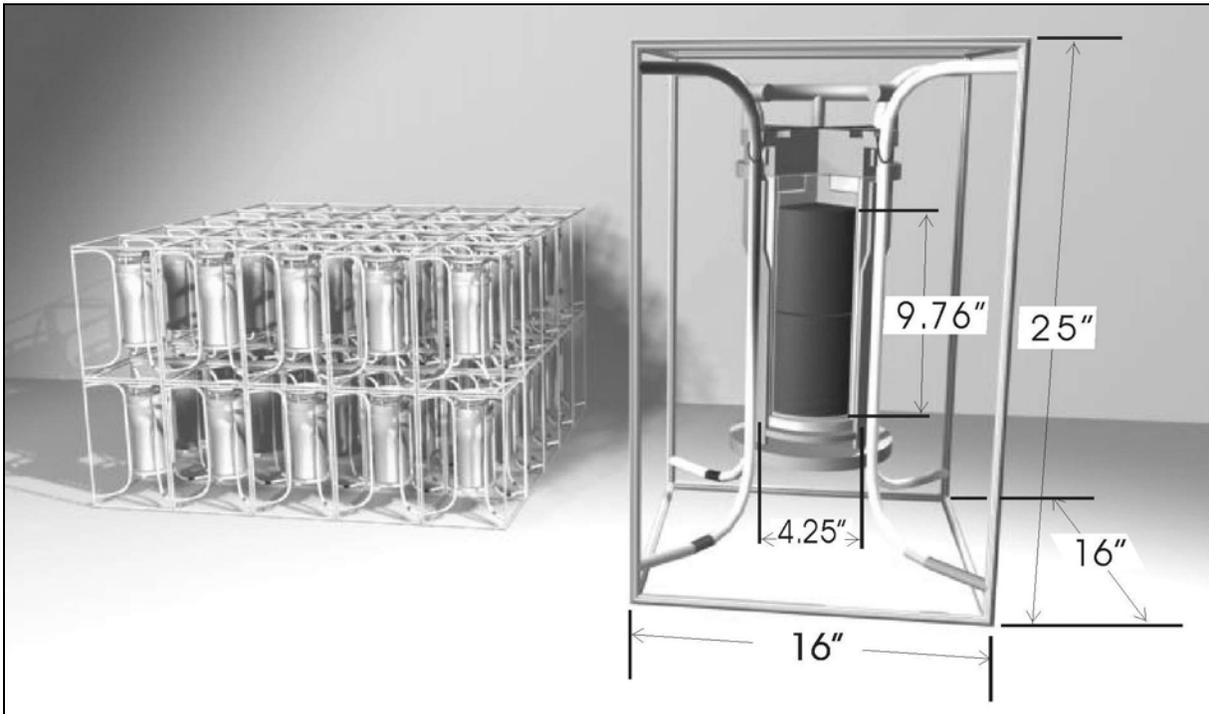


Figure 2. "Birdcage" container most likely used for storage and shipping of enriched uranium at the Huntington Pilot Plant. (Reproduced courtesy of S. Jensen from Paxton 1974)

To make a reasonable estimation of the potential dose to a worker from this material, it was assumed that a worker was located at 1 meter from the 5-by-5 double-stacked array for 2000 hours per year. This resulted in an annual exposure of 0.13 R per year. To estimate an upper 95<sup>th</sup> percentile exposure, it was assumed that the worker was located 1 foot from the stacked array for 2000 hours per year resulted in an annual exposure of 0.76 R per year.

The organ doses were calculated by multiplying the estimated annual exposure of 0.13 R by the "Exposure (R) to Organ Dose ( $H_T$ )" photon dose conversion factors found in Appendix B of the NIOSH

External Dose Reconstruction Implementation Guideline (DHHS 2002). The exposure geometry was assumed to be anterior-posterior (AP) and the dose rate was divided equally between photons with E=30-250 keV and photons with E>250 keV. Table 7 below shows the annual organ doses due to the potential exposure to the birdcage array.

Table 7. Annual organ doses due to exposure to a 5-by-5 array of birdcages.

Organ	Annual organ dose (rem)		
	Photons E=30-250 keV	Photons E>250 keV	Total
Bladder	8.09E-02	5.74E-02	1.38E-01
Red bone marrow	4.07E-02	4.68E-02	8.75E-02
Bone surface	7.99E-02	4.97E-02	1.30E-01
Breast	8.23E-02	6.05E-02	1.43E-01
Colon	6.89E-02	5.49E-02	1.24E-01
Esophagus	4.47E-02	4.84E-02	9.31E-02
Eye	8.03E-02	5.72E-02	1.38E-01
Ovaries	6.21E-02	5.32E-02	1.15E-01
Testes	9.32E-02	6.12E-02	1.54E-01
Liver	6.92E-02	5.49E-02	1.24E-01
Lung	6.41E-02	5.47E-02	1.19E-01
Remainder organs	5.71E-02	5.12E-02	1.08E-01
Skin	5.80E-02	5.43E-02	1.12E-01
Stomach	8.13E-02	5.75E-02	1.39E-01
Thymus	9.15E-02	5.80E-02	1.50E-01
Thyroid	9.36E-02	6.32E-02	1.57E-01
Uterus	6.57E-02	5.11E-02	1.17E-01

The organ doses in the second and third columns of Table 7 are entered into the NIOSH IREP program assuming a chronic exposure and a lognormal distribution with a GSD of 2.9. The organ doses in the second column are attributed to photons with E=30-250 keV and the organ doses in the third column are attributed to photons with E>250 keV.

The assumption was also made that workers received an annual occupationally related diagnostic x-ray. The exposure geometry was assumed to be posterior-anterior (PA) (DHHS 2002). The air kerma at skin entrance for the diagnostic chest x-ray was estimated to be 0.108 R (Scalsky 2003). The organ doses were calculated by multiplying the air kerma by the "Kerma ( $K_a$ ) to Organ Dose ( $H_T$ )" photon dose conversion factors for 30-250 keV photons found in Appendix B of the NIOSH External Dose Reconstruction Implementation Guideline (DHHS 2002). Table 8 below shows the annual organ doses due to the assumed annual diagnostic chest x-ray. The values in Table 8 are entered into the NIOSH-IREP program as the annual dose due to an acute exposure to photons (E=30-250 keV). The distribution is assumed to be normal with a standard deviation of 30%.

#### 4.0 Estimation of Exposure to Residual Activity

After conclusion of the AEC activities at the Huntington Pilot Plant sometime in 1963, the plant was shut down and maintained in stand-by condition (US DOE 1983). The site was then decontaminated between November 1978 and May 1979. All structures except a compressor building was removed and classified and contaminated scrap was buried at a classified location at the Portsmouth Plant in

Table 8. Annual organ doses due to the assumed annual diagnostic chest x-ray.

Organ	Annual dose (rem)
Bladder	0.074
Red bone marrow	0.124
Bone surface	0.178
Breast	0.059
Colon	0.095
Esophagus	0.105
Eye	0.022
Ovaries	0.110
Testes	0.061
Liver	0.101
Lung	0.133
Remainder organs	0.110
Skin	0.110
Stomach	0.077
Thymus	0.048
Thyroid	0.052
Uterus	0.096

Ohio. No results of the pre-decontamination survey could be found. According to a post-decontamination survey in 1980 (Berger 1981), exposure rates at 1 meter were between 9 and 12  $\mu\text{R/h}$  with a maximum of 35  $\mu\text{R/h}$ .

To estimate potential exposure to residual radioactivity from the end of AEC operations until decontamination of the site, the maximum exposure rate measured at the site in the 1980 survey was assumed to be the median exposure rate. The maximum exposure level recommended by the NCRP of 500 mrem per year (250  $\mu\text{R/h}$  for a 2000 hour work year) was assumed to be the upper 95<sup>th</sup> percentile exposure rate at the site. The annual organ doses from contaminated surfaces was calculated by multiplying the assumed median annual exposure of 70 mR (35  $\mu\text{R/h}$  for 2000 hours) by the Exposure (R) to Organ Dose (rem) photon dose conversion factors from Appendix B of the NIOSH External Dose Reconstruction Implementation Guideline (DHHS 2002). The exposure geometry was assumed to be isotropic and the annual exposure was divided evenly between the conversion factors for photons with energy between 30 and 250 keV and photons with energy greater than 250 keV. Table 9 shows the calculated annual organ doses from external exposure to residual radioactivity after the end of AEC operations at the site.

The organ doses in the second and third columns of Table 9 are entered into the NIOSH IREP program assuming a chronic exposure and a lognormal distribution with a GSD of 3.3. The organ doses in the second column are attributed to photons with  $E=30\text{-}250$  keV and the organ doses in the third column are attributed to photons with  $E>250$  keV. The residual contamination exposure period is assumed to begin on January 1, 1964 and end on May 31, 1979.

The results of the 1980 post-decontamination survey of the remaining compressor building were used to estimate potential exposure to residual activity after decontamination operations were completed at the site. To estimate the annual organ doses from contaminated surfaces in the plant, the median annual exposure of 24 mR (12  $\mu\text{R/h}$  for 2000 hours) was multiplied by the Exposure (R) to Organ Dose (rem) photon dose conversion factors from Appendix B of the NIOSH External Dose Reconstruction Implementation Guideline (DHHS 2002). The exposure geometry was assumed to be

Table 9. Annual organ doses due to external exposure to residual radioactivity from the end of AEC operations until decontamination.

Organ	Annual organ dose (rem)		
	Photons E=30-250 keV	Photons E>250 keV	Total
Bladder	1.88E-02	2.26E-02	4.14E-02
Red bone marrow	1.95E-02	2.33E-02	4.28E-02
Bone surface	3.28E-02	2.38E-02	5.67E-02
Breast	2.48E-02	2.59E-02	5.07E-02
Colon	1.80E-02	2.22E-02	4.02E-02
Esophagus	1.75E-02	2.29E-02	4.04E-02
Eye	2.60E-02	2.66E-02	5.25E-02
Ovaries	1.73E-02	2.19E-02	3.92E-02
Testes	2.21E-02	2.43E-02	4.64E-02
Liver	1.99E-02	2.33E-02	4.32E-02
Lung	2.19E-02	2.47E-02	4.66E-02
Remainder organs	1.94E-02	2.32E-02	4.26E-02
Skin	2.56E-02	2.66E-02	5.22E-02
Stomach	1.98E-02	2.32E-02	4.31E-02
Thymus	2.15E-02	2.39E-02	4.54E-02
Thyroid	2.24E-02	2.50E-02	4.74E-02
Uterus	1.70E-02	2.11E-02	3.81E-02

isotropic and the annual exposure was divided evenly between the conversion factors for photons with energy between 30 and 250 keV and photons with energy greater than 250 keV. Table 10 shows the calculated annual organ doses from external exposure to residual radioactivity after decontamination of the site.

Table 10. Annual organ doses due to external exposure to residual radioactivity after decontamination.

Organ	Annual organ dose (rem)		
	Photons E=30-250 keV	Photons E>250 keV	Total
Bladder	6.43E-03	7.76E-03	1.42E-02
Red bone marrow	6.68E-03	7.99E-03	1.47E-02
Bone surface	1.13E-02	8.17E-03	1.94E-02
Breast	8.50E-03	8.89E-03	1.74E-02
Colon	6.18E-03	7.61E-03	1.38E-02
Esophagus	6.00E-03	7.85E-03	1.38E-02
Eye	8.90E-03	9.11E-03	1.80E-02
Ovaries	5.93E-03	7.51E-03	1.34E-02
Testes	7.58E-03	8.32E-03	1.59E-02
Liver	6.82E-03	7.98E-03	1.48E-02
Lung	7.50E-03	8.47E-03	1.60E-02
Remainder organs	6.66E-03	7.96E-03	1.46E-02
Skin	8.77E-03	9.11E-03	1.79E-02
Stomach	6.79E-03	7.97E-03	1.48E-02
Thymus	7.37E-03	8.18E-03	1.56E-02
Thyroid	7.67E-03	8.57E-03	1.62E-02
Uterus	5.82E-03	7.25E-03	1.31E-02

The organ doses in the second and third columns of Table 10 are entered into the NIOSH IREP program assuming a chronic exposure and a lognormal distribution with a GSD of 1.9. The organ doses in the second column are attributed to photons with  $E=30-250$  keV and the organ doses in the third column are attributed to photons with  $E>250$  keV. The residual contamination exposure period is assumed to begin on June 1, 1979 and end on the employee's last day of work at the site.

The maximum internal exposure from residual radioactivity during both post-AEC time periods was estimated by assuming that the facility was uniformly contaminated at the level of maximum smear result (considered removable contamination) of 140 dpm/100 cm<sup>2</sup>. This value was multiplied by a resuspension factor of  $1E-06$  m<sup>-1</sup> (US NRC 2002c). This resulted in an estimated maximum residual air concentration of 0.006 pCi/m<sup>3</sup>. Assuming a breathing rate of 1.2 m<sup>3</sup>/h and a 2000-h work year results in a possible annual inhalation intake of 14 pCi. This value is considered negligible as it results in an annual dose of less than 1 mrem to the maximally exposed organ and is not included in the dose reconstruction.

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